LA-UR -81-1519

TITLE: SPECIAL SMALL-BREAK APPLICATIONS WITH TRAC

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SUBMITTED TO: American Nuclear Society
Specialists Conference on
Small Breaks in LWRs
Monterey, California
August 25-27, 1981

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DEPARTMENT OF ENERGY
CONTRACT W-7405-ENG. 36

SPECIAL SMALL-BREAK APPLICATIONS WITH TRAC

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ABSTRACT

Input models for the Transient Reactor Analysis Code (TRAC) are described and applications of these models to reactor transients involving small breaks in the primary coolant pressure boundary are demonstrated. The operation of the primary overpressure protection system (relief and safety valves) and the thermal-hydraulic response of the reactor to these transients are obtained from numerical simulations. Also, the effects of steam generator recirculation, steam generator tube rupture, Emergency Core Cooling (ECC) injection and reactivity feedback on the course and consequences of these transients are investigated. These models allow reliable predictions of accident signatures that can help determine the adequacy of equipment and procedures at nuclear power plants to prevent and to control severe accidents.

SPECIAL SMALL BREAK APPLICATIONS WITH TRAC

I. INTRODUCTION

The investigation of potentially severe Light-Water-Reactor (LWR) accidents through detailed computer simulation became possible with recent advances in computer technology and with development of best-estimate computer codes such as the Transient Reactor Analysis Code (TRAC).[1] The Los Alamos Severe Accident Sequence Analysis (SASA) effort takes advantage of these new capabilities to provide an important element in the analysis of such transients.

The SASA program involves identification of potential multiple-fault accidents (MFAs) and the thermal-hydraulic simulation of the primary and secondary system response to such accidents. The results will contribute to the Nuclear Regulatory Commission (NRC) evaluation of nuclear power plant preparedness for responding to these MFAs.

II. PLANT MODEL

Detailed, comprehensive studies of reactor operating experience led to the delineation of many potential MFAs, including those caused by operator actions,[2] which could lead to radiological releases from containment or to core damage. A computer model of the system was developed for accurately simulating the major systems of a pressurized-water reactor (PWR) and at the same time providing fast computational execution.

The initial investigations focused on a large four-loop Westinghouse PWR. A schematic of the TRAC model of this PWR is shown in Fig. 1. Three of the primary coolant loops were modeled as one combined loop to improve calculational efficiency, and the remaining loop containing the pressurizer was modeled separately. Included in the TRAC model of this reactor are numerical representations of the vessel, main coolant pumps, U-tube steam generators (with steam/water separation and recirculation), Emergency Core Cooling System (ECCS), primary volume control (letdown and makeup), hot and cold leg piping, pressurizer, pressurizer relief and safety valves (power operated and static check), pressure relief tank, secondary atmospheric relief valves and main steam lines. This model contains approximately 200 mesh cells or volumes for the finite-difference scheme employed by TRAC.[3]

III. SPECIFIC COMPONENT MODELS

Among the MFAs investigated at Los Alamos are small breaks in the pressurizer relief hardware (relief valves) and in the steam generator tube sheet. Simulation of these transients represents an application of TRAC not attempted previously. It therefore was necessary to modify existing component models and to include additional modeling capabilities in TRAC to handle the anticipated events during the transients.

Of major interest were the relief valves (located on the pressurizer) designed to prevent the primary pressure from exceeding the design value. Figure 2 is a schematic representation of this structure. The opening of these

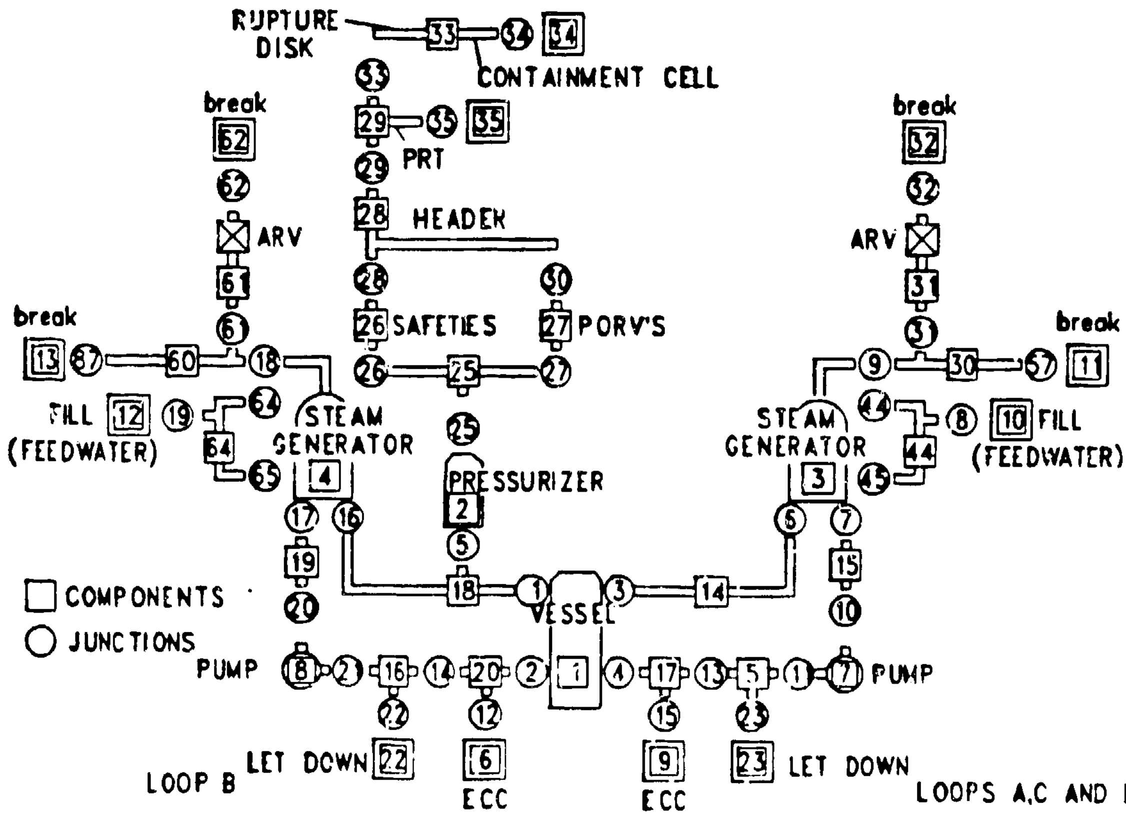

Fig. 1. TRAC schematic for four-loop Westinghouse PWR system.

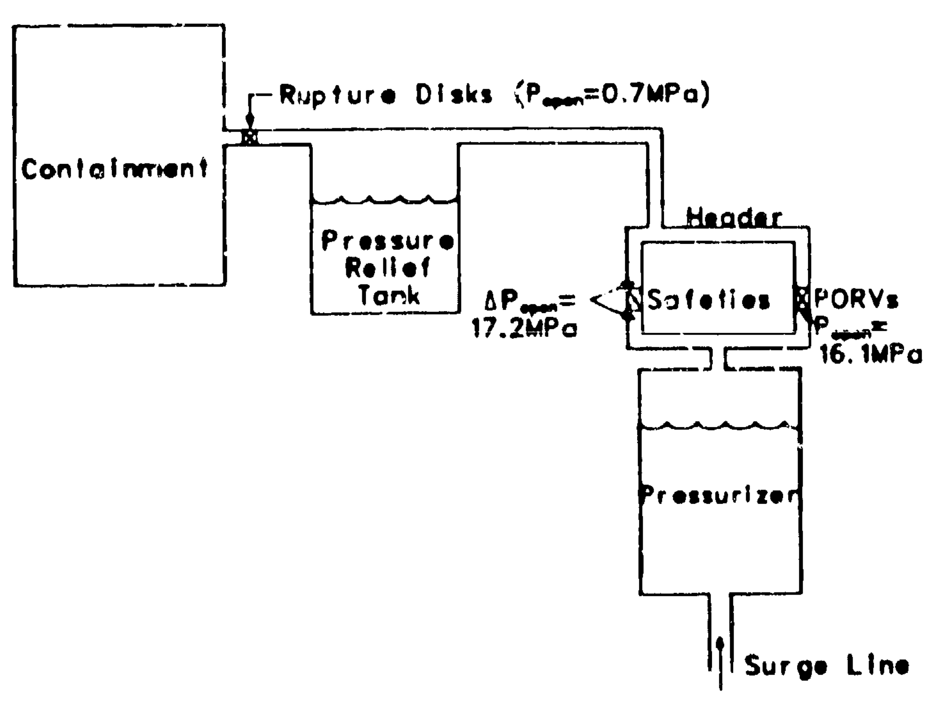

Fig. 2. Primary overpressure relief system used in TRAC model.

valves during a transient produces a small break in the primary system pressure boundary. Effluent from the relief valves is piped to the pressure relief tank (PRT), which then empties into the containment. Power-operated relief valves (PORVs) sense the upstream (pressurizer) pressure and open or close accordingly at specified rates. The valve options in TRAC were adapted to allow multiple pressure setpoints and opening and closing rates to simulate operation of these valves.

The safety relief valves (SRVs) are static check valves whose opening or closing is dependent upon the pressure gradient across the valve. Rapid fluctuations of this valve (flutter) and water hammer effects produced by rapid closing caused numerical instabilities. To alleviate these difficulties, a valve option was developed that allowed different opening and closing pressure-gradient setpoints along with different opening and closing rates. By using this option, the user can include hysteresis effects that produce an accurate physical representation of the valve while eliminating numerical instabilities.

PORV and SRV valves attached to the top of the pressurizer are joined to a common header leading to the PRT. Because the opening setpoint of the static check valve depends on the back pressure, it is important that the details of the header and PRT are modeled accurately. The PRT contains sufficient subcooled water to condense 110% of the pressurizer design discharge of steam while maintaining the tank pressure below 0.45 MPa. Once the tank pressure exceeds 0.69 MPa, rupture disks open providing a path to the reactor containment building. Many automatic plant actions (such as ECCS initiation and containment isolation) are dependent on the containment pressure and temperature, so considerable effort was necessary to model this primary overpressure protection system accurately. The details of the valves, header, PRT, and containment are necessary to analyze a small break in the primary steam space.

Anticipated transients without scram (ATWS) initiated by loss of feedwater to the steam generators are among the MFAs investigated at Los Alamos. These transients necessitate the opening of the PORVs and SRVs that feed back to the primary thermal-hydraulic response. During dryout of the steam-generator secondary water inventory, primary-to-secondary heat-transfer degradation occurs. Accurate prediction of this degradation requires that the recirculation and steam separation mechanisms of the U-tube steam generators be modeled. A schematic of this model is shown in Fig. 3. With accurate simulation of the recirculation flow, inlet and outlet conditions, liquid entrainment and water inventory, this model reliably predicted the heat transfer degradation, comparing well with results of more detailed computer codes designed expressly for analyzing steam-generator transient performance.[4] Steam generator tube ruptures also were modeled successfully at Los Alamos. Analyses were performed for steam-generator tube ruptures as the initiating events for transients and for ruptures induced by rapid steam-side depressurization. The primary and secondary sides of the steam generator were connected at the primary exit (cold leg), shown in Fig. 4, to represent a break at the tube sheet. The thermal-hydraulic characteristics of the tube representing the ruptures were selected to provide the correct flow losses due to friction within the U-tubes. Additional loss coefficients were included to account for form losses associated with the abrupt expansion at the tube sheet.

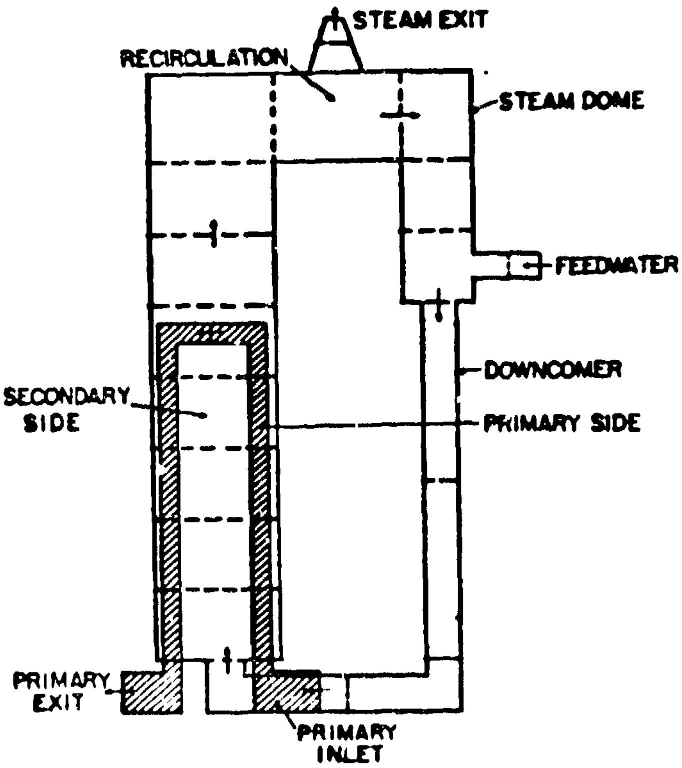

Fig. 3. Steam generator model with recirculation.

IV. MODEL APPLICATIONS

Since the incident at Three Mile Island, concern has risen regarding operational transients involving multiple failures of plant components and operator actions to control or mitigate accidents. The principal MFAs investigated at Los Alamos are those that result from loss of feedwater (LOFW) and include the effects of PORVs opening and closing (bleed-and-feed), ECCS unavailability, and failure to scram. Steam-generator tube ruptures (SGTR), including those involving a break in the main steam line, also have been investigated.

The temporary loss of main feedwater to the steam generators is an anticipated transient that happens a few times per year at many PWRs. Circumstances such as loss of off-site power, loss of electrical load, loss of condenser vacuum, as well as mechanical failure of pumps or valves can cause loss of main feedwater. When feedwater flow is lost, the potential for a severe accident exists if feedwater cannot be recovered before the residual liquid in the steam generator secondaries is lost. TRAC's numerical simulations of feedwater transients evaluate the effectiveness of automatic safety functions and define strategies for controlling and mitigating such accidents. At the initiation of the feedwater transient, the four-loop PWR is assumed to lose main feedwater flow to all steam generators when auxiliary feedwater is unavailable; all automatic plant safety systems except emergency

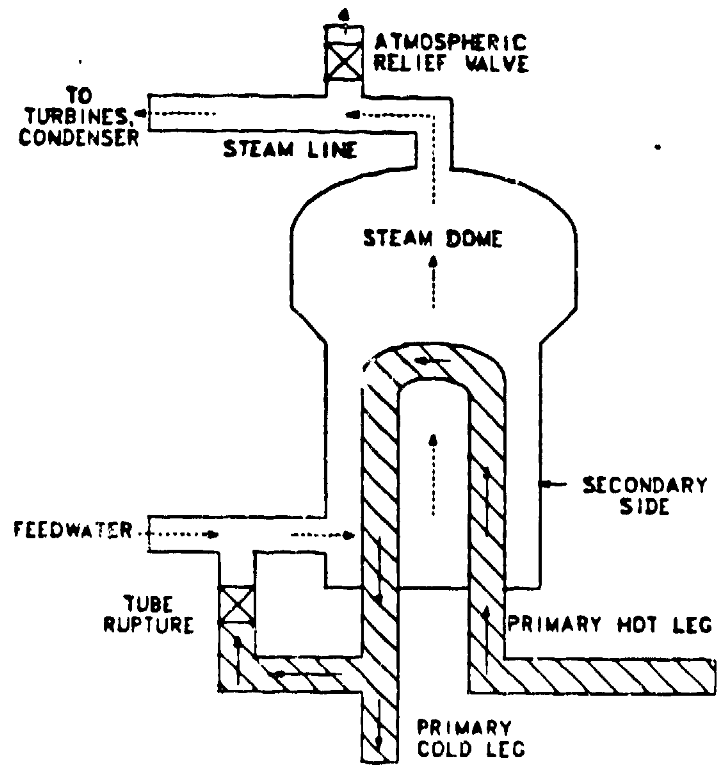

Fig. 4. Steam generator tube rupture model.

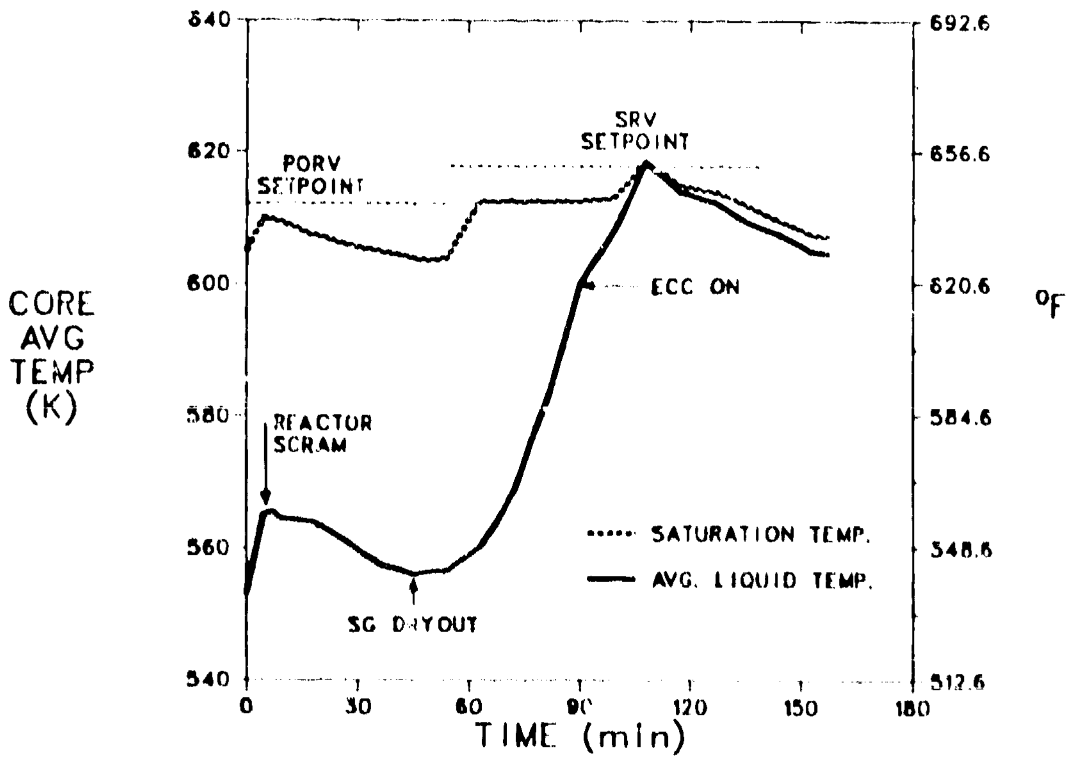

Fig. 5. Safety systems function to prevent core damage.

feedwater are assumed to be operational. The sequence for this transient is indicated in Fig. 5, illustrating the nominal response to the LOFW transient. A reactor scram signal was generated by the reduced feedwater flow, and the control rods were inserted with a nominal shutdown margin of 3 1/2% δk/k. The primary temperature rose briefly from the initial undercooling of the core but then decreased as the reactor power decayed. After the steam generator liquid inventory boiled off, the primary pressure and temperature began rising because of loss of heat sink. The pressure rose to the setpoint of the PORVs, which opened to accommodate the subcooled expansion of the primary coolant. The PORV flow area was sized to produce the rated steam flow at design conditions. It should be noted that the PORV flow area during the transient was controlled by the expansion rate of the primary, so that the PORV opened only enough to relieve this expansion. Shortly after the PORVs opened, the rupture disks on the PRT broke, releasing steam into the containment building, and the pressure in this building began rising. The ECCS was actuated by a containment overpressure signal, and the slope of the primary coolant temperature (Fig. 5) decreased. Primary coolant expansion could not be accommodated by flow out the PORVs, so the SRVs opened briefly. Saturation conditions were then reached and boiling began. The primary coolant continued to boil until the power decayed such that the ECCS flow could remove all the decay energy produced.

When recovery of emergency feedwater flow is not expected for one hour or longer, bleed-and-feed operations using the PORVs and the ECCS offer a viable recovery mode. Figure 6 shows a series of actions involving the PORVs and the atmospheric relief valves (ARVs) that would allow the operators to lower the

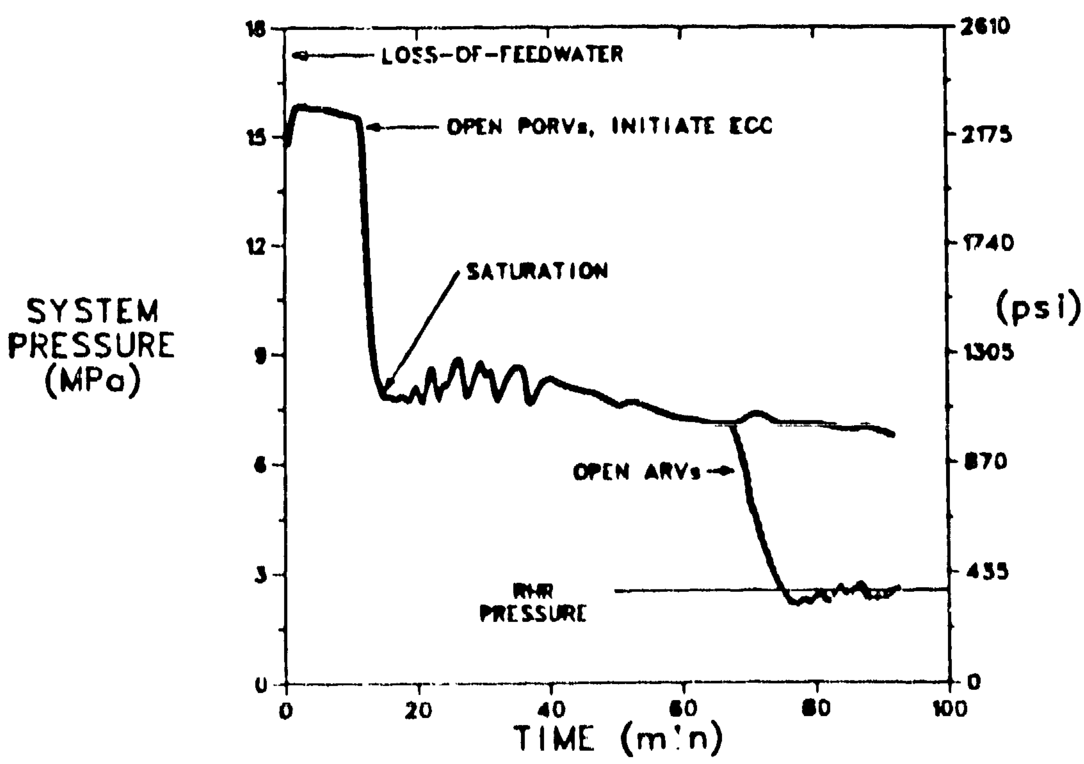

Fig. 6. Bleed and feed demonstrated as a viable recovery mode.

system pressure below the design limits of the residual heat removal system (3.0 MPa). If the steam generator secondaries empty before the ARVs are opened manually, the core can remain covered as long as ECCS water is supplied.

Another application of the PWR TRAC model is the loss-of-main-feedwater accident coupled with failure to insert control rods and failure of an opened PORV to close. This accident uses all the features of this TRAC model, including the effect of ECCS boron content on the reactor power. Figure 7 shows the response of the system during this ATWS. As the steam generator secondary-side water level decreased, the primary temperature slowly rose. This resulted in a decrease in reactor power from the negative reactivity feedback associated with increasing primary coolant temperatures. When the steam generators emptied, the primary side heated up more rapidly and the power level decreased to about 6% of the initial power level. During this time, the PORVs and SRVs have cycled completely open to relieve the primary coolant expansion. When the containment overpressure signal initiated the ECCS, the primary coolant temperature began decreasing and the power started leveling. The auxiliary feedwater and ECC flows were sufficient to remove the reactor heat, so the SRVs and PORVs closed. However, one PORV is assumed to fail in the fully open position, allowing the primary system pressure and temperature to continue decreasing. The lowered pressure allowed greater ECC flow, and the increased reactivity from system cooldown was balanced by decreased reactivity due to boron addition. Thus, the system reached a quasi-equilibrium state. This transient demonstrates the complex and interrelated feedback associated

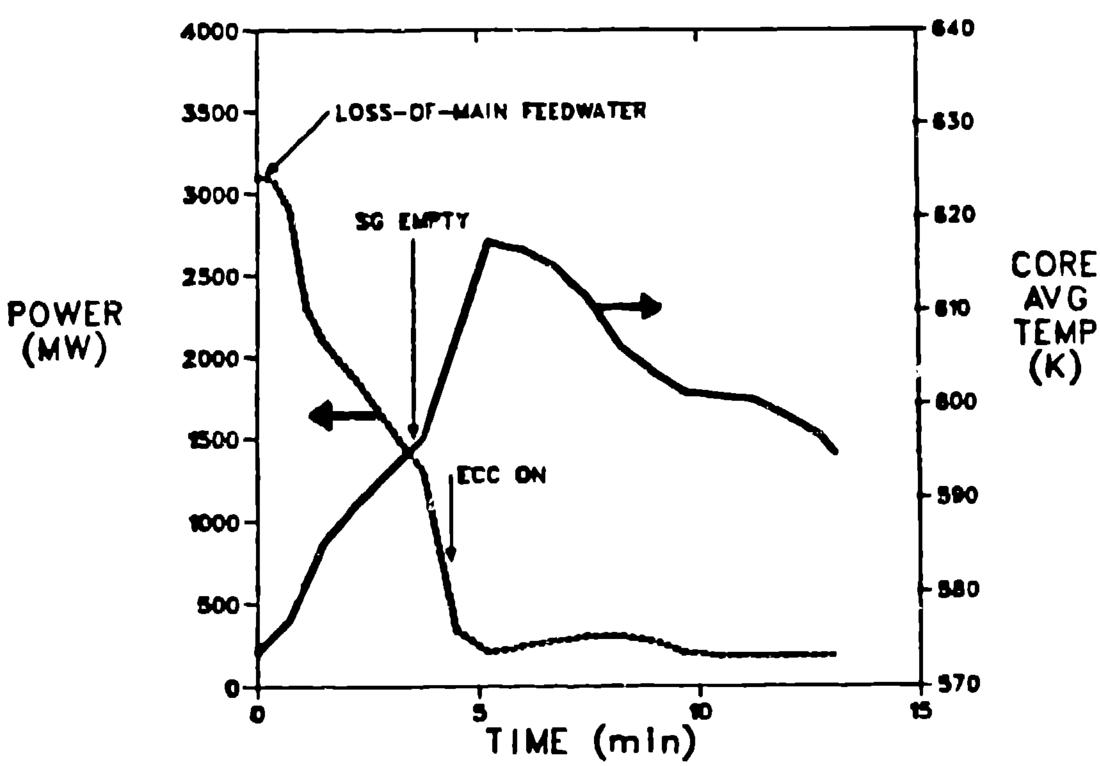

Fig. 7. Steam generator dryout results in rapid power shutdown.

with steam-generator dryout, injection of borated water, and pressure and temperature changes associated with valve opening.

The model for steam-generator tube ruptures allowed the investigation of the primary-secondary response during such events. A transient was postulated in which a double-ended break of the loop B main steam line occurred, inducing the rupture of between one and five tubes of that steam generator. The secondary side of the damaged loop rapidly depressurized and a reactor trip signal was generated by the excess steam flow. This rapid depressurization, along with the primary coolant leakage into the secondary, resulted in a rapid drop in the primary pressure as shown on Fig. 8. Low primary pressure activated the ECCS and initiated recovery. The ECCS flow rate, which depends on primary pressure, equilibrated with the leakage flow and prevented core uncovery. The ECC flow compensated for the leakage flow through a maximum of five ruptured tubes. Hot primary liquid was replaced with subcooled ECCS liquid, so the core cooled and the primary system remained subcooled throughout the transient.

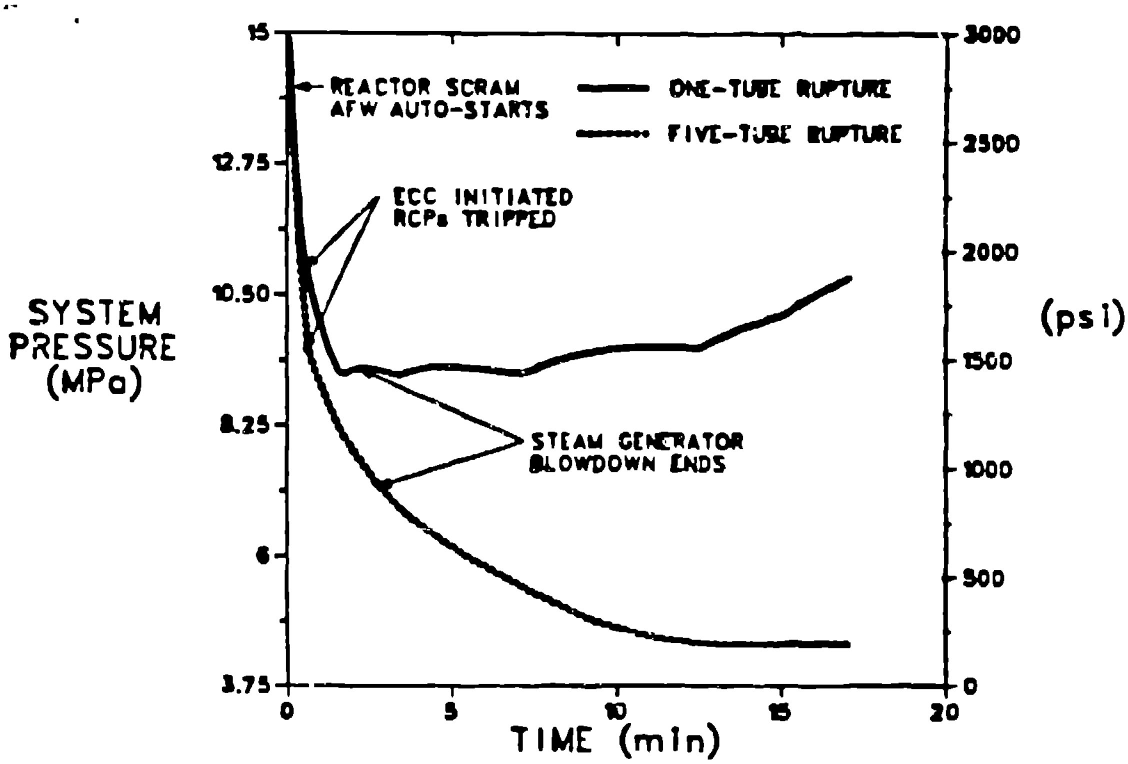

Fig. 8. System pressure indicates plant response.

V. CONCLUSIONS

These applications of the TRAC PWR model demonstrate the advanced analysis capability available for small-break transients. Complex component systems can be coupled easily, along with operational trip logic, to model a diverse set of reactor systems.

Many of these transients can be calculated analytically, at least in part. However, the complex feedback mechanisms of the total system response cannot be handled adequately in this way. A comprehensive systems code such as TRAC can give the user valuable insight into these complex interactions and can predict unanticipated or unexpected synergistic effects. As additonal mechanisms and interactions are identified, the models can be modified to reflect this behavior. The investigation of these transients, therefore, provides the NRC and the nuclear industry with information necessary for evaluating whether safety systems and operating procedures are adequate for controlling severe accidents.

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